TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

John A. Scalice
Vice President, Watts Bar Site Operations

NOV 1 8 1997

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of the	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 FACILITY OPERATING LICENSE NPF-90 - LICENSEE EVENT REPORT (LER) 50-390/97015

The purpose of this letter is to provide LER 50-390/97015. This LER resulted from a manual reactor trip due to decreasing steam generator levels. These conditions are being reported in accordance with 10 CFR 50.73(b)(2)(iv). The enclosure provides this LER.

If you should have any questions, please contact P. L. Pace at (423) 365-1824.

Sincerely,

R T Purcell for

J. A. Scalice

Enclosure

1/1

IE22

cc (Enclosure):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. Robert E. Martin, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

CATEGORY 1

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SCALICE,J.A. Tennessee Valley Authority
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SUBJECT: Forwards LER 97-015-00,re manual reactor trip due to decreasing steam generator levels.Enclosure listed.

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20603.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Watts Bar Nuclear Plant - Unit 1	05000390	1 OF 8

TITLE (4)

MANUAL REACTOR TRIP DUE TO FEEDWATER ISOLATION

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
10	19	97	97	015	00	11	18	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 12

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract
20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. N. Mays, Licensing Engineer	(423)-365-3855

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 19, 1997 at 0950 EDT during Unit 1 startup, the reactor was in Mode 1, approximately 12% power. The turbine was at 1800 RPM and preparing for connection to the grid. Steam generator levels began decreasing due to isolation of the inservice feedwater heaters. Auxiliary feedwater was manually initiated and reactor power was manually decreased to 4% using control rods. The reactor was manually tripped due to decreasing steam generator levels. The feedwater heaters isolated on high-high level. The main control room alarms for high feedwater heater levels did nót alarm. All other systems responded to the initiating event as expected and within analyzed parameters. There were no abnormal radiological conditions throughout the event. The cause of the event was isolation of the high pressure feedwater heater strings due to high level with a contributing cause of failure of the high level alarms in the heaters to provide annunciation in the control room. The high level in the A1 and B1 heaters was due to a failure to recognize that additional operator attention was required to control the level in the A1 and B1 heaters with the C1 heater isolated, during turbine roll and main generator synchronizing. The procedure was revised to caution operators on the impact of feedwater heaters out of service during this period. The high-high alarm level switches on the feedwater heaters were repaired and verified to be operable.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. PLANT CONDITIONS:

On October 19, 1997, at 0950 EDT, the plant was in Mode 1, Reactor Coolant System (RCS) (EIIS AB) average temperature was 559°F, RCS pressure was 2240 psig.

II. DESCRIPTION OF EVENT

A. Event

On October 19, 1997 at 0950 EDT during Unit 1 startup, the reactor was in Mode 1, approximately 12% power. The turbine was at 1800 RPM and preparing for connection to the grid. Level began to rise in the A1 and B1 feedwater heaters (HX) during the turbine roll as extraction steam began to condense on the shell side of the heaters. Level increased in the high pressure heater shells until automatic heater isolation occurred. Steam generator (SG) levels began decreasing due to isolation of the inservice feedwater heaters. Auxiliary feedwater system (AFW) (BA) was manually initiated and reactor power was manually decreased to 4% using control rods. The reactor was manually tripped due to decreasing steam generator levels. All systems responded as required with the following exception: The 1A main feedwater pump (MFP) failed to receive a trip signal following main feedwater isolation signal. The 1A MFP was isolated and not in service at that time but was being prepared for service, (the standby feedwater pump was in service). The feedwater heaters isolated on high-high level. The main control room alarms (LA) for high feedwater heater levels did not alarm.

All other systems responded to the initiating event as expected and within analyzed parameters. There were no abnormal radiological conditions throughout the event.

Problem Evaluation Report (PER) WBPER971290 was initiated to document this event in the TVA Corrective Action Program.

B. Inoperable Structures, Components, or Systems that Contributed to the Event

The high level indication in both the A1 and B1 heaters did not initiate the alarms in the main control room.

C. Dates of Discovery and Reportable Findings

10/19/97 Prior to the event	C1 and C2 feedwater heaters are out of service. C1 was available to be returned to service. Maintenance was working on C2.

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II. DESCRIPTION OF EVENT (continued)

C. Dates of Discovery and Reportable Findings (continued)

04:30	System Engineer walks feedwater heater system down and discusses with shift Operations the need to lower the level in the heaters.
07:00	The A1 and B1 feedwater heaters had been restored to normal operating levels based on sight glass.
07:15	Turbine Roll began.
09:45 (approx)	A1 and B1 feedwater heaters isolated due to high level.
09:46:11	Standby main feedwater pump indicates low flow.
09:46:55	Steam generator levels low.
09:47:56	Auxiliary feedwater pumps manually started. Reactor power manually runback to approximately 4%.
09:48:41	Steam generator levels low-low.
09:50:07	Reactor manually tripped.

D. Other Systems or Secondary Functions Affected

No other systems or secondary functions were affected.

E. Method of Discovery

The event was monitored through control room indication of the steam generator levels by Operations personnel as it occurred.

F. Operator Actions

Operations personnel manually tripped the reactor.

G. Automatic and manual safety system responses

The manual initiation of the AFW to maintain steam generator levels, the manual reactor trip and subsequent turbine trip, occurred as designed. The RCS responded as expected to the reactor trip.

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II. DESCRIPTION OF EVENT (continued)

G. Automatic and manual safety system responses (continued)

Following the reactor trip, a feedwater isolation occurred as required. The 1A MFP failed to receive a trip signal on the feedwater isolation signal; however, these pumps were not in service at the time, and steam to the MFP turbines were secured, preventing flow. Lifted leads associated with an ongoing work activity were identified as the cause of this occurrence. The leads were subsequently landed and the circuit returned to normal.

III. CAUSE OF EVENT

The cause of the event was isolation of the high pressure heater strings due to high level in the A1 and B1 heaters. The high level in the A1 and B1 heaters was due to a failure to recognize that additional operator attention was required to control the level in the A1 and B1 heaters with the C1 heater isolated, during turbine roll and main generator synchronizing.

Contributing Causes:

1. The failure of the high-level alarms in the A1 and B1 feedwater heaters contributed to the event by failing to provide advance warning to the main control room operators that a high level was reached.

2. A misunderstanding of a modification on piping arrangement in the #2 feedwater heaters that was performed during the mid-cycle outage contributed to an inadequate level of monitoring during startup.

3. C1 heater being out of service increased the need for additional attention of the heaters due to the more rapid filling of the A1 and B1 heaters.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES

A. Evaluation of Plant Systems /Components

Reactor Coolant System/Reactor Core Response

The reactor core and RCS responded as expected to the reactor trip. Prior to the event, reactor power was approximately 12-14%. Due to operator action reactor power was approximately 4% at the time of the trip.

Condensate/ feedwater

Initial Conditions: Prior to the event, the condensate / feedwater system was aligned to support startup of the unit in preparation of synchronizing the turbine generator to the grid. Hotwell pumps B and C and the Standby MFP were in operation. The MFPs were secured. The main feedwater

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IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES (continued)

A. Evaluation of Plant Systems/Components (continued)

regulating valves were in manual operation for forward flush and the bypass regulating valves were in automatic controlling steam generator level. The #3 and #7 heater drain tank pumps were not in service. The C1 high pressure feedwater heater was not in service. The C2 feedwater heater was out of service for maintenance.

Feedwater Heaters: At the initiation of the event, level began to rise in the A1 and B1 feedwater heaters during the turbine roll as extraction steam began to condense on the shell side of the heaters. Due to the design of the #1 feedwater heaters, pressures in the heaters must exceed 3.25 psi to drain the water from the heaters. There is minimal pressure to drain the water in the heater during this phase of startup operation. With the C1 heater out of service, level increased at a more rapid rate than previous startups. (Heater levels are not indicated in the control room). The higher heater level was not identified and increased to the high level alarm point in the heater. However, the high level alarms in both the A1 and B1 heaters failed to initiate a control room alarm and the A1 and B1 heaters isolated on high-high level, resulting in a loss of flow to the steam generators. Level began to decrease in the steam generators. Operators started the two motor driven AFW pumps and the turbine driven AFW pump to recover level and began reducing reactor power using control rods. Level continued to decrease and control room operators manually tripped the reactor. Following the reactor trip and initiation of AFW, steam generator levels continued to decline briefly, then began to return to normal levels. The brief continuing decline in level was due to steam generator shrink and operator action limiting the flow to the steam generators to prevent overcooling the RCS in accordance with Reactor Trip Response procedures.

The piping arrangement of the #1 feedwater heaters is such that when exposed to vacuum conditions from the main condenser on the downstream shell side and minimal extraction steam pressure on the upstream shell side, there is limited force to push the water forward to the #2 heater or the main condenser prior to synchronizing the generator. The drain line for the heater is at a level of about 10 feet above the center line of the heater. Prior to turbine roll, level in the heater must be maintained as low as possible to make space for the condensed extraction steam. During turbine roll, close monitoring of levels in the heaters is needed to adjust the opening of the condenser bypass valve, and to open the startup vent valve when necessary to use atmospheric pressure to push the water up the drain line. With the C1 heater isolated, additional extraction steam would be drawn to the A1 and B1 heaters, making the level more sensitive. The combination of the high drain elevation and the increased flow due to the C1 heater being isolated caused the level problem in the A1 and B1 heaters.

Feedwater Isolation Signal: Following the reactor trip, a feedwater isolation occurred as required. The 1A MFP failed to receive a trip signal on the feedwater isolation signal. The MFPs were not in service at the time, and steam to the MFP turbines were secured, preventing flow. Lifted leads associated with an ongoing work activity were identified as the cause of this occurrence. The leads were subsequently landed and the circuit returned to normal.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES (continued)

B. Evaluation of Personnel Performance

In response to plant status, the Operations personnel manually tripped the reactor, which initiated a turbine trip. Operators responded in accordance with Emergency Operating Instructions E-0, Reactor Trip; ES-01, Reactor Trip Response; AOI-17, Turbine Trip and AOI-34, Immediate Boration. The prompt action of the operators was consistent with plant protection with manual initiation of AFW, manual reactor trip, and manual control of plant cooldown.

C. Safety Significance

The loss-of-feedwater transient experienced at WBN on October 19, 1997, is bounded by the FSAR Loss-of-Normal Feedwater described in Section 15.2.8. The loss-of-feedwater results in a reduced ability to remove heat generated in the reactor core. Protection systems are provided in the design to prevent the loss of heat sink from allowing core heatup and damage. These protective systems included:

1. Reactor Trip on low-low water level in any steam generator.
2. Automatic start of AFW motor driven pumps on low-low level in any one generator, trip of both turbine driven MFPs, safety injection signal, and loss of offsite power.
3. Automatic start of the turbine driven AFW pump on low-low level in any two generators, trip of both turbine driven MFPs, safety injection signal, and loss of offsite power.

These features restore the ability of the system to provide adequate core cooling by reducing the heat produced in the reactor core (reactor trip) to decay heat values and providing a guaranteed minimum safety grade AFW flow for the steam generators to remove heat. Analyses were performed for this event demonstrating the above protective features are adequate to prevent over pressurization of the RCS and loss of water from the reactor core. Conservatism in the FSAR analysis include:

1. Steam generator levels at initially low values
2. Plant operating at 102% power
3. Steam generator heat transfer coefficients consistent with natural circulation
4. Failure of the turbine driven AFW supply
5. Maximum AFW supply temperature
6. Steam generator steam relief via safety valves
7. High initial RCS average temperature and pressurizer level
8. Steam generator low-low level trip assumed to be 0% narrow range (58% wide range)

The actual plant transient experienced is less severe in that steam generator initial water levels were normal for startup power levels, plant power was approximately 12-14%, offsite power was not lost during the event, steam generator heat transfer was consistent with normal RCS flow, AFW was

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IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES (continued)

C. Safety Significance (continued)

manually started and operated normally, AFW temperature was not at the high limit, condenser dumps were available and operated, RCS initial conditions were nominal for startup conditions, and a manual reactor trip was initiated before a low-low level alarm occurred on any steam generator. Response of the plant is therefore, bounded by the FSAR event. The loss-of-feedwater event did not challenge the pressurizer safety valves by water discharge and did not result in core cooling problems. The event was properly handled by the operators. Due to the low decay heat and low initial power levels, the RCS temperature transient was dominated by cooldown and was controlled by operator manual action. The prompt action of the operators was consistent with plant protection with manual initiation of AFW, manual reactor trip, and manual control of plant cooldown.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Operations personnel inserted control rods to reduce reactor power to 4% and manually initiated AFW pumps. The reactor was manually tripped in response to decreasing steam generator levels.

B. Corrective Actions to Prevent Recurrence

Revise System Operating Instruction (SOI) 5&6.1, "Extraction Steam, Heater Drains, and Vent System," as needed to control feedwater heater levels between turbine roll and main generator synchronizing. Include a caution to ensure awareness of the effects of heaters not in service on the remaining heaters.

Develop training for placing the number 1 and 2 feedwater heaters in service in the event of a forced outage. This training will be covered during normally scheduled requalification cycles.

Brief Operations and Technical Support personnel on this event. Include examples of infrequently performed tasks and request feedback to identify additional examples similar to the this issue.

Verify high level switches on feedwater heaters 1, 2, and 4 (A, B, and C) are functioning properly and repair as necessary. Verify proper function of high-high level switches on A1, B1, and C1 feedwater heaters.

VI. ADDITIONAL INFORMATION

A. Failed components

1. Safety Train Inoperability

There were no safety trains inoperable.

VI. ADDITIONAL INFORMATION (continued)

A. Failed components (continued)

2. Component/System Failure Information

a. Method of Discovery of Each Component or System Failure:

There were no component/system failures due to this event.

b. Failure Mode, Mechanism, and Effect of Each Failed Component:

There were no component/system failures due to this event.

c. Root Cause of Failure:

There were no component/system failures due to this event.

d. For Failed Components With Multiple Functions, List of Systems or Secondary Functions Affected:

There were no component/system failures due to this event.

e. Manufacturer and Model Number of Each Failed Component:

There were no component/system failures due to this event.

B. Previous Similar Event

A review of previous similar events at WBN was conducted. Although several events for WBN Unit 1 have been reported for loss-of-feedwater that resulted in the reactor being tripped, no events which isolated the inservice heaters due to high-high levels with similar root causes have been reported under 10 CFR 50.72 or 10 CFR 50.73.

VII. COMMITMENTS

Implementation of the actions tabulated in Section V, Corrective Actions has been completed.